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Abstract

Automatic control of the average core flux by measuring N^{16} coolant activity at the FRM Munich Research Reactor

by

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Stability of the neutron flux very often is a prerequisite for exact results of experiments using a reactor as a neutron source. Usually the flux is measured at the reflector of the core in order to control the total flux. As a result the flux at the chamber position is kept constant, but this leaves a possibility open for a variation of the average core flux by as much as 20 %, caused by a change of the flux distribution in and around the core.

To avoid this the average core flux of the FRM is kept constant by a new cascade control system using N^{16} coolant activity. The N^{16} activity depends on the fission rate in the core and exceeds all the rest of the activities of the coolant by far. It may be determined independent of small variations about the mean value of the water flow rate. A suitable arrangement will be described where the N^{16} activity depends only on the average core flux. The dependance is linear; the control signal, however, is delayed by about 10 sec with respect to the variations of the flux in the core. The gamma radiation of the N^{16} activity is determined

by measuring the current of an ionization chamber. The overall accuracy is 0.25 %. This current is averaged over a period of 8 sec. Its deviation from a demand current drives a fast auxiliary circuit. This fast circuit consists of a chamber near the core and a continuous regulating system operating the control rod. Fast variations of the neutron flux are levelled out by the auxiliary circuit, the average core flux is kept constant by the N^{16} circuit. The observation of thermal power as well as flux determinations for varying flux distribution in the core proved the stability of the desired average core flux even for long periods.

As a result of a constant average core flux the neutron beam intensity averaged over all beam tubes has a minimum of variations. A special arrangement and operation of the shim rods will be devised to still further decrease these variations. An optimal solution to this problem was theoretically found and proved by flux measurements in the core using an U^{235}-Thermopile.

The motion of the control rod should be limited to a small range about a fixed start position in order not to vary the flux distribution in the core appreciably. The control rod position is therefore tested every 15 sec. In case its positional change has already exceeded this limit, the shim rods are automatically moved in small steps until the control rod reaches its starting position again.

The discussed control circuits based at the N^{16} activity proved to be very successful, e.g. for intermittent reactor operation the variation of intensity of the P3 neutron beam always stayed below 1.8 % over a period of several weeks, for continuous operating at constant power of 1 MW the variation is lower as 0.7 % compared to former variations of 12 % at the same beam tube.

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Abstract

DETERMINATION OF l_{eff} AND β_{eff} IN A CRITICAL REACTOR BY THE PULSED NEUTRON TECHNIQUE

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The values of l_{eff} and β_{eff} for a critical reactor were determined by the pulsed neutron technique. The method is based on the analysis of the rise and the decay of the thermal neutron density produced by injection of a fast neutron pulse in the reactor.

The response of an essentially homogeneous reactor to a square fast neutron pulse of intensity S_1 and duration Δ is given by:

$$n_1(0 \leqslant t \leq \Delta) = n_o + SBt + S \sum_{j=1}^{6} A_j (e^{a_j t} - 1) \quad \text{and} \tag{1}$$

$$n_1(t \geqslant \Delta) = n_o + SB\Delta + S \sum_{j=1}^{6} A_j e^{a_j t} (1 - e^{-a_j \Delta}), \tag{2}$$

where

$n_1(t)$ is the thermal neutron density for the first pulse,

n_o is the thermal neutron density at $t=0$,

$$B = 1/\left(1 + \frac{1}{l_{eff}} \sum_{i=1}^{6} \frac{\beta_i}{\lambda_i}\right) \tag{3}$$

$$A_j = \prod_{i=1}^{6} (a_j + \lambda_i) / a_j^2 \prod_{i \neq j} (a_j - a_i) \qquad j = 1, 2, \ldots 6, \tag{4}$$

a_j are the roots of the equation

$$1 + \frac{1}{l_{eff}} \sum_{i=1}^{6} \frac{\beta_i}{s + \lambda_i} = 0, \tag{5}$$

S is proportional to S_1,

and a_6 is the most negative root of equation (5) and characterizes the prompt neutron decay.

Equation (1) reduces to

$$n_1(0 \leqslant t \leqslant \Delta) = n_o + S(B+1)t \qquad (6)$$

for $|a_j t| \ll 1$, taking into account that $A_6 \cdot a_6 \simeq 1$

The thermal neutron density during the time of the m^{th} neutron pulse in m consecutive pulses of period $T \gg 1/|a_6|$ will be

$$n_m\left[(m-1)T \leqslant t \leqslant (m-1)T+\Delta\right] = n_o + (m-1)SB\Delta + S(B+1)t \qquad (7)$$

where $(m-1)SB\Delta$ takes into account the rise of the power level due to the previous (m-1) pulses.

In the conduct of the experiment the accumulation of m consecutive pulses gives a number proportional to:

$$n(t)=n_1+n_2+\ldots+n_m=mn_o+\frac{m(m-1)}{2}SB\Delta+mS(B+1)t \qquad (8)$$

In analyzing the experimental results it is noted that

a) the slope X of the straight line obtained from the experimental plot of equation (8) is proportional to $mS(B+1)$ and

b) the measurement of "initial" (mn_o) and "final" ($mn_o+\frac{m(m+1)}{2}SB\Delta$) background gives a quantity Y proportional to $\frac{m(m+1)}{2}SB\Delta$.

Combining the measured quantities X and Y one obtains l_{eff} as a function of $\beta_{i,eff}$ using the known values of λ_i. The decay constant a_6 can be determined directly as the most negative root of equation (5). The values of $\beta_{i,eff}$ may be adjusted so that a_6 calculated above coincides with the measured one from the decay of the pulse.

The measuring equipment consists of a 14 MeV pulse neutron generator, a 512 channel multiscaler and an automatic device to preset the number of pulses.

This method has been used to determine the values of l_{eff} and β_{eff} for the "Democritos" reactor.

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Abstract

POWER LEVEL TEMPERATURE COEFFICIENT BY ZERO POWER MEASUREMENTS

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The reactivity effects of temperature changes in a research or power reactor under various operating conditions is usually determined by experimental measurements. This generally requires a series of experiments in which each type of operating mode is explored. A simple experimental method is described here in which zero power experiments permit evaluation of temperature coefficients applying to any mode of power range operation.

The method has been used for measurements in a pool reactor but it could be applied to other types of reactors as well where provisions are made for circulating the coolant. Basically the experiment involves the measurement of the single cell temperature coefficient $\alpha_i(r,T)$ for the various locations in the core and reflector, corresponding to a uniform temperature T.

Measurements of $\alpha_i(r,T)$ for each cell are obtained by separately heating this particular cell, while the remaining cells are held at a constant temperature by operation of the coolant circulating system.

From this data the temperature effects for the following conditions can then be evaluated.

A. Isothermal Temperature Coefficient: The average isothermal temperature coefficient can be expressed in terms

of the single cell coefficients since

$$\alpha_{total} = \sum_{i=1}^{n} \alpha_{i,c}(r,T_u) + \sum_{j=1}^{m} \alpha_{j,ref}(r,T_u)$$

where:

$\alpha_{i,c}(r,T_u)$ is the average temperature coefficient of the i^{th} cell of the core ($i = 1,2....n$).

$\alpha_{j,ref}(r,T_u)$ is the average temperature coefficient of the j^{th} cell of the reflector ($j=1,2...n$).

T_u is constant.

B. Non-isothermal Temperature Coefficient: The average temperature coefficient of a reactor operating at power levels sufficient to cause non-isothermal temperature conditions is given by

$$\alpha_{total} = \sum_{i=1}^{n} \alpha_{i,c}(r,T_s) + \sum_{j=1}^{m} \alpha_{j,ref}(r,T_u)$$

where:

T_s depends on the cell position.

This response implies that the temperature change is uniform throughout the reactor core and reflector as would be caused by a small change in coolant inlet temperature.

C. Non-uniform Temperature Coefficient: The temperature response for a non-uniform temperature change may also be described in terms of the single cell coefficients. For cases where the power level is changed while flow rate is held constant or vice versa, the temperature rise will vary in proportion to the flux distribution so that

$$\alpha_{total} = \frac{1}{\bar{\Phi}} \sum_{i=1}^{n} \alpha_{i,c}(r,T_s)\Phi(r)$$

where:

$\frac{\Phi(r)}{\bar{\Phi}}$ is the lateral flux distribution.

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Abstract

LIGHT WATER ENRICHED URANIUM LATTICE EXPERIMENTS

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Light water moderated 10 % enriched uranium fuelled lattices have been investigated with different H/U^{235} ratios. One of the aims of these experiments was to test the results of calculation methods used. For this purpose the measurements were carried out in a wide interval of the H/U^{235} ratio in order to get significant effects.

Fuel elements of the VVR-S type research reactor were used and in most cases triangular lattice patterns applied.

Critical mass values were determined over the range 190 - 775 of the H/U^{235} ratio. This range involves nearly equal lengths of undermoderated and overmoderated regions.

Temperature dependence of reactivity was determined in the interval 10 - 40 ^{o}C, in overmoderated, undermoderated and in a nearly optimum lattice. Temperatures of fuel and moderator were the same. In this temperature interval both positive and negative values of temperature coefficients were found in all the lattices.

Spatial dependence of the reactivity effect of vertical, cylindrical voids - the lengths of which were

identical with that of the fuel elements - was determined in four cores with different lattice pitches. The values of the void to cell volume ratios were kept nearly the same. Positive void effect was reached in a strongly overmoderated lattice.

Thermal neutron distributions were determined in lattices containing central water traps of different diameters. These measurements were carried out both in a strongly undermoderated and in a nearly optimum lattice.

The value of β^{eff}/l was measured using both the Feynmann and the Rossi method in a nearly optimum lattice. The two methods yielded nearly the same results. The value of β^{eff}/l was determined in an overmoderated and an undermoderated lattice by the Rossi method only - which proved to be faster and more accurate.

The results of a simple calculation method were compared with the experimental data. Rather good agreement was found for the two group model when taking into account fission under slowing down and the intracell dependence of the thermal spectrum in the calculation of the thermal utilisation factor.

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THERMAL NEUTRON CROSS SECTIONS AND DIFFUSION PARAMETERS IN SEVERAL ORGANIC MATERIALS

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In the past few years considerable effort has been spent on measuring the influence of chemical binding on the scattering of slow neutrons in several hydrogenous liquids. This is important from practical point of view since the spectrum shape of neutrons in hydrogenous moderators depends on the binding effect. The quantitative understanding of neutron thermalization requires a fairly good knowledge of the scattering law for slow neutrons in hydrogenous liquids.

Unfortunately the determination of the scattering law involves rather large experimental difficulties. The measurement of the integral scattering cross section is simpler but it is not so sensitive to the details of the scattering system. In spite of this the knowledge of the integral cross sections may have importance in the calculation of the transport cross section. The temperature

dependent transport cross section can be obtained from the integral scattering cross section using the Radkowsky prescription which is rather useful but theoretically not rigorous.

The paper gives a detailed analysis of the Radkowsky prescription on the base of integral cross section and diffusion parameter measurements in several organic liquids /benzene, cyclohexane, n-hexane, toluene, xylene, and diphenyl/. The energy dependence of the integral neutron cross sections was measured by time of flight technique and calculated on the base of Zemach-Glauber's theory. The calculated data are in good agreement with the experimental results. The diffusion parameters were obtained from the measurements carried out with the pulsed neutron source method. The average transport cross section obtained from the cross section measurements was compared with the results of the pulsed source measurements. The comparison shows that using the Radkowsky prescription the calculated values of the transport cross sections are in fairly good agreement with those of the pulsed source measurements.

The transport cross sections were also calculated

from the data of $\sigma_s^H(E)$ for 1,3-butadiene since Petric et al. proposed the use of these data for the calculation of the transport cross sections in various hydrocarbons. Unfortunately the 1,3-butadiene data do not give satisfactory results, therefore the knowledge of the energy dependent cross section seems to be important in each of the investigated hydrocarbons for the calculation of the average transport cross section.

As a conclusion it should be emphasized that the Radkowsky prescription is to be considered rather as a useful tool, not as a theoretically rigorous treatment of the neutron scattering in moderators.

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Abstract

STATISTICAL THEORY OF NEUTRON CHAIN REACTIONS

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A statistical theory is developed for the chain reactions in nuclear reactors. The probability distribution function /PDF/ for the number of neutrons in a given state is derived with assumptions of quite general character. From the PDF the fundamental equation for the generating function is obtained permitting the calculation of the expectation value and the variance of the neutron density. Equations are derived for the PDF of the number of fissions and neutrons counted during a given time-interval. Several limit theorems are derived for the generating functions. These theorems are important in various applications of the theory. The multi-group approximation is introduced into the framework of the statistical theory.

In the paper several applications are illustrated. The fluctuation of the neutron yield in pulsed fast reactors is considered and the theory of the prompt neutron period measurements in thermal reactors is developed on the base of neutron density fluctuations.

In the former case calculations are made in one-group approximation and the time dependence of the neutron density

fluctuations is computed for different reactivity changes. A simple form of time-dependence on reactivity is chosen. The reactivity of the system at the start of the run is assumed to be negative. Reactivity then increases to a definite positive value, and decreases to its original negative value. It is shown that the number of neutrons and fissions in a simple burst fluctuates strongly even if the average number of pulse-initiating neutrons is relatively high.

In the latter case the effect of the delayed neutrons on the accuracy of measurements is analysed and the correlation between the numbers of neutrons counted in consecutive time intervals is examined. Spatial fluctuations of neutron density are also studied and on the base of this study a new method is proposed for the criticality parameter measurements. As a result of theoretical considerations it is shown that the correlation method can be used in the case of thermal reactors too, provided the average neutron lifetime is not larger than about 10^{-4} sec. The theoretical basis of the "Zero Probability Method" is given and a mathematical procedure is derived for the determination of the prompt period with help of this method.

Finally a review is given on the latest results of the statistical theory of neutron chain reactions.

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Abstract

A Study on the Diffusion Length of Thermal Neutrons in Terphenyl (Santowax - R) at High Temperatures up to 400°C

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It is needless to say that the temperature dependence of the diffusion length is quite important in relation with the neutron transport cross section as a basic reactor parameter according to the temperature rise of modern reactors.

The diffusion length of thermal neutrons in various moderators containing hydrogen has been studied for a long time, especially in the theoretical field. However, only few experiments continuously in the wide temperature range have been reported, although some experiments and calculations at the temperatures comparatively low or at the spot temperature have been published.

It seems that this is due to the engineering difficulties, such as using water at high temperature.

However, it is being thought that the deviation from the 1/v law due to the effect of molecular binding would be fairly remarkable at the temperature above 200°C.

In this work the temperature dependence of diffusion length of thermal neutrons in terphenyl ($C_{18}H_{14}$) has been measured in the wide range from 200°C to 400°C continuously, and the transport cross section was studied in connection with the results. From the results in comparison with the calculated value, using $\sigma_S(E)$ from Brugger's experimental data on the scattering law about Santwax-R, it is clear that there is still a problem in assuming that the diffusion length varies as $\frac{1}{\rho}\sqrt{T}$ at high temperature over 300°C, and it

might be considered that the temperature dependence of the diffusion length in hot water would be deduced from an analogy of this work.

In this report many engineering problems as mentioned below, encountered in making the measurements under this condition, will be stated, too.

(1) Uniformity of temperature distribution geometrically during the experiments.

(2) The figure of the flux including the effect of the epithermal neutrons.

(3) Measurement of the neutron flux, that is the difficulty of using cadmium and indium foils due to the high temperature, i.e., development of gadolinium and dysprosium foils.